NUREG-0800
(Formerly NUREG-75/087)

U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

6.2.1.3 MASS AND ENERGY RELEASE ANALYSIS FOR POSTULATED LOSS-OF-COOLANT ACCIDENTS

REVIEW RESPONSIBILITIES

Primary - Containment Systems and Severe Accident Branch (SCSB)[1]

Secondary - None

I. AREAS OF REVIEW

The SCSB[2] reviews the analyses of the mass and energy release to assure that the data used to evaluate the containment and subcompartment functional design are acceptable for that purpose. The review includes the following areas:

1. The energy sources that are available for release to the containment.

2. The mass and energy release rate calculations for the initial blowdown phase of the accident.

3. For pressurized water reactor (PWR) plants, because of the additional steam generator stored energy available for release, the mass and energy release rate calculations for the core reflood and post-reflood phases of the accident.

Review Interfaces[3]

In addition, the SCSB will coordinate other branches' evaluations that interface with the overall review of this area as follows: The Mechanical Engineering Branch (EMEB)[4] is responsible for reviewing the acceptability of piping design criteria, selected break locations and break sizes

USNRC STANDARD REVIEW PLAN

Standard review plans are prepared for the guidance of the Office of Nuclear Reactor Regulation staff responsible for the review of applications to construct and operate nuclear power plants. These documents are made available to the public as part of the Commission's policy to inform the nuclear industry and the general public of regulatory procedures and policies. Standard review plans are not substitutes for regulatory guides or the Commission's regulations and compliance with them is not required. The standard review plan sections are keyed to the Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants. Not all sections of the Standard Format have a corresponding review plan.

Published standard review plans will be revised periodically, as appropriate, to accommodate comments and to reflect new information and experience.

Comments and suggestions for improvement will be considered and should be sent to the U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, Washington, D.C. 20555.

based on the provisions made to limit pipe motion, for breaks postulated to occur within subcompartments ~~(see~~ as part of its primary review responsibility for Standard Review Plan Section 3.6.2~~)~~.

For that area of review identified above as being reviewed as part of the primary review responsibility of another branch, the acceptance criteria and their methods of application are contained in the referenced SRP section.

II. ACCEPTANCE CRITERIA

The acceptance criteria given below ~~applies~~ apply[5] to the mass and energy release analysis for postulated loss-of-coolant accidents. SCSB[6] accepts the mass and energy analysis if the relevant requirements of General Design Criterion 50 and 10 CFR Part 50, Appendix K, paragraph I.A are complied with. The relevant requirements are as follows:

A. General Design Criterion 50, as it relates to the containment and subcompartments[7] being designed with sufficient margin, requires that the containment and its associated systems can accommodate, without exceeding the design leakage rate, and the containment and subcompartment design can withstand the calculated pressure and temperature conditions resulting from any loss-of-coolant accident.

B. 10 CFR Part 50, Appendix K, as it relates to sources of energy during the LOCA, provides requirements to assure that all the energy sources have been considered.

In meeting the requirements of General Design Criterion 50 the following specific criterion or criteria that pertain to the mass and energy analysis are used as included below:

1. Sources of Energy

The sources of stored and generated energy that should be considered in analyses of loss-of-coolant accidents include: reactor power; decay heat; stored energy in the core; stored energy in the reactor coolant system metal, including the reactor vessel and reactor vessel internals; metal-water reaction energy; and stored energy in the secondary system (PWR plants only), including the steam generator tubing and secondary water.

Calculations of the energy available for release from the above sources should be done in general accordance with the requirements of 10 CFR Part 50, Appendix K, paragraph I.A~~(Ref. 2)~~[8]. However, additional conservatism should be included to maximize the energy release to the containment during the blowdown and reflood phases of a LOCA.

The requirements of paragraph I.B in Appendix K to 10 CFR Part 50, concerning the prediction of fuel clad swelling and rupture should not be considered. This will maximize the energy available for release from the core.

2. Break Size and Location

a. The staff's review of the applicant's choice of break locations and types is discussed in SRP Section 3.6.2.

b. Of several breaks postulated on the basis of a., above, the break selected as the reference case for subcompartment analysis should yield the highest mass and energy release rates, consistent with the criteria for establishing the break location and area.

c. Containment design basis calculations should be performed for a spectrum of possible pipe break sizes and locations to assure that the worst case has been identified.

3. Calculations

In general, calculations of the mass and energy release rates for a loss-of-coolant accident should be performed in a manner that conservatively establishes the containment internal design pressure (i.e., maximizes the post-accident containment pressure and the containment subcompartment response). The criteria given below for each phase of the accident indicate the conservatism that should exist.

a. Subcompartment Analysis

The analytical approach used to compute the mass and energy release profile will be accepted if both the computer program and volume noding of the piping system are similar to those of an approved emergency core cooling system (ECCS) analysis. The computer programs that are currently acceptable include SATAN-V (Reference~~.18~~ 28)[9], CRAFT-2 (Reference~~.17~~ 27)[10], CE FLASH-4 (Reference~~.19~~ 16)[11], and RELAP4 (Reference~~.15~~ 3)[12], when a flow multiplier of 1.0 is used with the applicable choked flow correlation. An alternate approach, which is also acceptable, is to assume a constant blowdown profile using the initial conditions with an acceptable choked flow correlation.

b. Initial Blowdown Phase Containment Design Basis

The initial mass of water in the reactor coolant system should be based on the reactor coolant system volume calculated for the temperature and pressure conditions existing at 102% of full power~~(Ref. 2)~~[13].

Mass release rates should be calculated using a model that has been demonstrated to be conservative by comparison to experimental data.

Calculations of heat transfer from surfaces exposed to the primary coolant should be based on nucleate boiling heat transfer. For surfaces exposed to steam, heat transfer calculations should be based on forced convection.

Calculations of heat transfer from the secondary coolant to the steam generator tubes for PWRs should be based on natural convection heat transfer for tube surfaces immersed in water and condensing heat transfer for the tube surfaces exposed to steam.

c. PWR Core Reflood Phase (Cold Leg Breaks Only)

Following initial blowdown of the reactor coolant system, the water remaining in the reactor vessel should be assumed to be saturated. Justification should be provided for the refill period. An acceptable approach is to assume a water level at the bottom of the active core at the end of blowdown so there is no refill time.

Calculations of the core flooding rate should be based on the emergency core cooling system operating condition that maximizes the containment pressure either during the core reflood phase or the post-reflood phase.

Calculations of liquid entrainment; i.e., the carryout rate fraction, which is the mass ratio of liquid exiting the core to the liquid entering the core, should be based on the PWR FLECHT experiments (Reference~~.20~~ 29)[14]. Liquid entrainment should be assumed to continue until the water level in the core is 61 cm (2 feet)[15] from the top of the core. An acceptable approach is to assume a carryout rate fraction (CRF) of 0.05 to the 46 cm (18-inch)[16] core level, a linearly increasing CRF to 0.80 at the 61 cm (24-inch)[17] level, and a constant CRF of 0.80 until the water level is 61 cm (2 feet)[18] from the top of the core. Above this level, a CRF of 0.05 may be used.

The assumption of steam quenching should be justified by comparison with applicable experimental data. Liquid entrainment calculations should consider the effect on the carryout rate fraction of the increased core inlet water temperature caused by steam quenching assumed to occur from mixing with the ECCS water.

Steam leaving the steam generators should be assumed to be superheated to the temperature of the secondary coolant.

d. PWR Post-Reflood Phase

All remaining stored energy in the primary and secondary systems should be removed during the post-reflood phase.

Steam quenching should be justified by comparison with applicable experimental data.

The results of post-reflood analytical models should be compared to applicable experimental data.

e. PWR Decay Heat Phase

The dissipation of core decay heat should be considered during this phase of the accident. The fission product decay energy model is acceptable if it is equal to or more conservative than the decay energy model given in Branch Technical Position ASB 9-2 in SRP Section 9.2.5.

Steam from decay heat boiling in the core should be assumed to flow to the containment by the path which produces the minimum amount of mixing with ECCS injection water.

The following computer models are acceptable for calculating mass and energy releases for containment design basis calculations: the Westinghouse model (SATAN-V)~~(Ref. 18)~~[19], the B&W model (CRAFT-2)~~(Ref. 17)~~[20], the C.E. model (CE FLASH-4)~~(Ref. 19)~~[21], and the G.E. blowdown model (Reference~~.~~ 23)[22]. Other methods will be acceptable if they are found by SCSB[23] to be conservative for these calculations.

Technical Rationale[24]

The technical rationale for application of the above acceptance criteria to the mass and energy release analysis for postulated loss-of-coolant accidents is discussed in the following paragraphs.

1. GDC 50 requires the containment structure and associated heat removal system to be designed with margin to accommodate any loss-of-coolant accident such that the containment design leak rate is not exceeded. A loss-of-coolant accident potentially causes the greatest pressure surge and release of fission products when compared to any other accident. Since it is the most severe challenge expected, containment must be designed to definitively withstand this accident. Following GDC 50 will ensure that containment integrity is maintained under the most severe accident conditions thus precluding the release of radioactivity to the environment.

2. Appendix K to 10 CFR 50 provides required and acceptable features of evaluation models used to analyze various circumstances applicable to the ECCS. Section I.A of Appendix K provides a comprehensive list of LOCA heat (energy) sources and the reactor operating history assumptions associated with those heat sources. Since the mass and energy release analysis for postulated loss-of-coolant accidents is used to design containment and containment subcompartments such that they will withstand the worst case LOCA, it is critical that all potential energy sources are taken into account. Following 10 CFR 50 Appendix K will ensure that containment and containment subcompartments are designed to accommodate all energy sources for the worst case

LOCA, thus precluding the potential release of radioactivity to the environment following such a LOCA.

III. REVIEW PROCEDURES

The procedures described below are followed for the review of the mass and energy release analysis for loss-of-coolant accidents. The reviewer selects and emphasizes material from these procedures as may be appropriate for a particular case. Portions of the review may be carried out on a generic basis or by applying the results of previous reviews of similar plants.

The SCSB confirms,[25] with the EMEB, the validity of the applicant's analysis of pipe break size, type and locations for subcompartments containing high energy lines by using elevation and plan drawings of the containment showing the routing of lines containing high energy fluids. The SCSB[26] determines that an appropriate reference case for subcompartment analysis has been identified. In the event a pipe break other than a double-ended pipe rupture is postulated by the applicant, the EMEB[27] will evaluate the applicant's justification for assuming a limited displacement pipe break.

The SCSB compares the sources of energy considered in the loss-of-coolant analysis and the methods and assumptions used to calculate the energy available for release from the various sources with the acceptance criteria listed in section II, above. The SCSB determines the acceptability of the analytical models and the assumptions used to calculate the rates of mass and energy release during the initial blowdown, core reflood, and post-reflood phases of a loss-of-coolant accident. The SCSB[28] also compares energy inventories at various times during a loss-of-coolant accident to ensure that the energy from the various sources has been accounted for and has been transferred to the containment on an appropriate time scale.

The SCSB[29] reviews comparisons made by the applicant to experimental data and makes comparisons to other available experimental data to determine the amount of conservatism in the mass and energy release models.

The SCSB[30] may perform confirmatory analyses of the mass and energy profiles. The purpose of the analysis is to confirm the predictions of the mass and energy release rates appearing in the safety analysis report, and to confirm that an appropriate break location has been considered in these analyses.

For standard design certification reviews under 10 CFR Part 52, the procedures above should be followed, as modified by the procedures in SRP Section 14.3 (proposed), to verify that the design set forth in the standard safety analysis report, including inspections, tests, analysis, and acceptance criteria (ITAAC), site interface requirements and combined license action items, meet the acceptance criteria given in subsection II. SRP Section 14.3 (proposed) contains procedures for the review of certified design material (CDM) for the standard design, including the site parameters, interface criteria, and ITAAC.[31]

IV. EVALUATION FINDINGS

The conclusions reached on completion of the review of this SRP section are presented in SRP Section 6.2.1.

V. IMPLEMENTATION

The following is intended to provide guidance to applicants and licensees regarding the NRC staff's plan for using this SRP Section.

This SRP section will be used by the staff when performing safety evaluations of license applications submitted by applicants pursuant to 10 CFR 50 or 10 CFR 52.[32] Except in those cases in which the applicant proposes an acceptable alternative method for complying with specified portions of the Commission's regulations, the method described herein will be used by the staff in its evaluation of conformance with Commission regulations.

The provisions of this SRP section apply to reviews of applications docketed six months or more after the date of issuance of this SRP section.[33]

VI. REFERENCES

The references for this SRP section are listed in Standard Review Plan Section 6.2.1.

[This Page Intentionally Left Blank]

SRP Draft Section 6.2.1.3

Attachment A - Proposed Changes in Order of Occurrence

Item numbers in the following table correspond to superscript numbers in the redline/strikeout copy of the draft SRP section.

Item	Source	Description
1.	Current PRB names and abbreviations	Editorial change made to reflect current PRB name and responsibility for this SRP Section.
2.	Current PRB names and abbreviations	Editorial change made to reflect current PRB name and responsibility for this SRP Section.
3.	SRP-UDP format item, Reformat Areas of Review	Added "Review Interfaces" heading to Areas of Review. Reformatted existing description of review interface to describe how SCSB reviews aspects of mass and energy release under other SRP sections and how other branches support the review.
4.	Current PRB names and abbreviations	Editorial change made to reflect current PRB name and responsibility for SRP Section 3.6.2.
5.	Editorial	A clarification was made by changing "applies" to "apply" which is the correct form of the word in this sentence.
6.	Current PRB names and abbreviations	Editorial change made to reflect current PRB name and responsibility for this SRP Section.
7.	Editorial	"Subcompartment" was changed to "subcompartments" to reflect the fact that a containment may have more than one subcompartment.
8.	SRP-UDP format item	Format change to make the citation of references consistent with SRP-UDP requirements.
9.	SRP-UDP format item/Unverified reference	Format change to make the citation of references consistent with SRP-UDP requirements. The reference number was changed to be consistent with revisions to the Reference section in SRP 6.2.1. This reference cannot be verified to be the most current reference that is still approved by the NRC.
10.	SRP-UDP format item/Unverified reference	Format change to make the citation of references consistent with SRP-UDP requirements. The reference number was changed to be consistent with revisions to the Reference section in SRP 6.2.1. This reference cannot be verified to be the most current reference that is still approved by the NRC.

Item	Source	Description
11.	SRP-UDP format item/Unverified reference	Format change to make the citation of references consistent with SRP-UDP requirements. The reference number was changed to be consistent with revisions to the Reference section in SRP 6.2.1. This reference cannot be verified to be the most current reference that is still approved by the NRC. The NRC approved CE's use of CEFLASH4A in the System 80+ FSER (see PI 24427). However, CEFLASH4A is not the same version of this computer code that is cited in SRP 6.2.1.3.
12.	SRP-UDP format item/Unverified reference	Format change to make the citation of references consistent with SRP-UDP requirements. The reference number was changed to be consistent with revisions to the Reference section in SRP 6.2.1. This reference cannot be verified to be the most current reference that is still approved by the NRC.
13.	SRP-UDP format item	Format change to make the citation of references consistent with SRP-UDP requirements.
14.	SRP-UDP format item/Unverified reference	Format change to make the citation of references consistent with SRP-UDP requirements. The reference number was changed to be consistent with revisions to the Reference section in SRP 6.2.1. This reference cannot be verified to be the most current reference that is still approved by the NRC.
15.	SRP-UDP format item, Metrication policy implementation	The existing criteria of 2 ft for core water level below the top of the core at which liquid entrainment is assumed to stop was converted to 61 cm using the guidance of Federal Standard 376B. See enclosed conversion worksheet.
16.	SRP-UDP format item, Metrication policy implementation	The existing criteria of 18 inches for the core level at which to discontinue assuming a .05 carryout rate fraction was converted to 46 cm using the guidance of Federal Standard 376B. See enclosed conversion worksheet.
17.	SRP-UDP format item, Metrication policy implementation	The existing criteria of 24 inches for the core level at which to discontinue assuming a linearly increasing carryout rate fraction was converted to 61 cm using the guidance of Federal Standard 376B. See enclosed conversion worksheet.
18.	SRP-UDP format item, Metrication policy implementation	The existing criteria of 2 ft for the level from the top of the core at which to discontinue assuming a .80 carryout rate fraction was converted to 61 cm using the guidance of Federal Standard 376B. See enclosed conversion worksheet.

SRP Draft Section 6.2.1.3

Attachment A - Proposed Changes in Order of Occurrence

Item	Source	Description
19.	SRP-UDP format item/Unverified reference	Format change to make the citation of references consistent with SRP-UDP requirements. This reference cannot be verified to be the most current reference that is still approved by the NRC. Additionally, the name of the vendor computer code was added for clarity and for consistency with section II.B.3.a.
20.	SRP-UDP format item/Unverified reference	Format change to make the citation of references consistent with SRP-UDP requirements. This reference cannot be verified to be the most current reference that is still approved by the NRC. Additionally, the name of the vendor computer code was added for clarity and for consistency with section II.B.3.a.
21.	SRP-UDP format item/Unverified reference	Format change to make the citation of references consistent with SRP-UDP requirements. This reference cannot be verified to be the most current reference that is still approved by the NRC. The NRC approved CE's use of CEFLASH4A in the System 80+ FSER (see PI 24427). However, CEFLASH4A is not the same version of this computer code that is cited in SRP 6.2.1.3. Additionally, the name of the vendor computer code was added for clarity and for consistency with section II.B.3.a.
22.	SRP-UDP format item/Unverified reference	Format change to make the citation of references consistent with SRP-UDP requirements. This reference cannot be verified to be the most current reference that is still approved by the NRC.
23.	Current PRB names and abbreviations	Editorial change made to reflect current PRB name and responsibility for this SRP Section.
24.	SRP-UDP format item, Develop Technical Rationales	Added Technical Rationale for GDC 50 and 10 CFR 50, Appendix K. Technical Rationale is a new SRP-UDP format item.
25.	Editorial	A comma was added to clarify the sentence.
26.	Current PRB names and abbreviations	Editorial change made to reflect current PRB name and responsibility for this SRP Section (2 identical changes in this paragraph).
27.	Current PRB names and abbreviations	Editorial change made to reflect current PRB name and responsibility for SRP Section 3.6.2 (2 identical changes in this paragraph).
28.	Current PRB names and abbreviations	Editorial change made to reflect current PRB name and responsibility for this SRP Section (3 identical changes in this paragraph).
29.	Current PRB names and abbreviations	Editorial change made to reflect current PRB name and responsibility for this SRP Section.

Item	Source	Description
30.	Current PRB names and abbreviations	Editorial change made to reflect current PRB name and responsibility for this SRP Section.
31.	SRP-UDP Guidance, Implementation of 10 CFR 52	Added standard paragraph to address application of Review Procedures in design certification reviews.
32.	SRP-UDP Guidance, Implementation of 10 CFR 52	Added standard sentence to address application of the SRP section to reviews of applications filed under 10 CFR Part 52, as well as Part 50.
33.	SRP-UDP Guidance	Added standard paragraph to indicate applicability of this section to reviews of future applications.

SRP Draft Section 6.2.1.3

Attachment B - Cross Reference of Integrated Impacts

Integrated Impact No.	Issue	SRP Subsections Affected
	No Integrated Impacts were incorporated in this SRP Section.	